UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
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KING OF PRUSSIA, PA 19406-1415

December 19, 2011

Mr. Michael J. Pacilio
Senior Vice President, Exelon Generation Company, LLC
President and Chief Nuclear Officer, Exelon Nuclear
4300 Winfield Rd.
Warrenville, IL 60555

SUBJECT: LIMERICK GENERATING STATION – NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS TEAM INSPECTION REPORT 05000352/2011007 AND 05000353/2011007

Dear Mr. Pacilio:

On November 4, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Limerick Generating Station (LGS), Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on November 4, 2011, with Mr. Peter Gardner, Plant Manager, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. In conducting the inspection, the team reviewed selected procedures, calculations and records, observed activities, and interviewed station personnel.

This report documents one NRC-identified finding of very low safety significance (Green). This finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the finding was entered into your corrective action program, the NRC is treating the finding as a non-cited violation (NCV), consistent with Section 2.3.2 of the NRC's Enforcement Policy. If you contest the NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Limerick Generating Station. In addition, if you disagree with the cross-cutting aspect of the finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Resident Inspector at the Limerick Generating Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system, Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

Lawrence T. Doerflein

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

Docket Nos. 50-352; 50-353
License Nos. NPF-39; NPF-85

Enclosure:
Inspection Report 05000352/2011007; 05000353/2011007
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

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Sincerely,

/RA/

Lawrence T. Doerflein, Chief
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REGION I

Docket Nos.:	50-352, 50-353
License Nos.:	NPF-39, NPF-85
Report Nos.:	05000352/2011007 and 05000353/2011007
Licensee:	Exelon Generation Company, LLC
Facility:	Limerick Generating Station, Units 1 and 2
Location:	Sanatoga, PA 19464
Inspection Period:	October 17 through November 4, 2011
Inspectors:	K. Mangan, Senior Reactor Inspector, Division of Reactor Safety (DRS), Team Leader C. Williams, Reactor Inspector, DRS J. Rady, Reactor Inspector, DRS
Approved By:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000352/2011007, 05000353/2011007; 10/17/2011-11/04/2011; Limerick Generating Station Units 1 and 2; Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications.

This report covers a two week on-site inspection period of the evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by three region based engineering inspectors. One finding of very low risk significance (Green) was identified, which was considered to be a non-cited violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a non-cited violation of 10 CFR 50.63, "Loss of All Alternating Current (AC) Power," because Exelon did not demonstrate that the alternate AC (AAC) source could provide acceptable capability to withstand a station blackout (SBO) within the analyzed coping timeline. Specifically, Exelon's evaluation of the Limerick Generating Station's excess emergency diesel generator (EDG) capacity did not analyze the effects of the loss of an operating emergency service water (ESW) pump following a single failure on the non-blacked out unit. The loss of the ESW pump would result in loss of cooling to one of the three credited EDGs and a subsequent high temperature trip of the EDG. The team determined the time delay to reset this trip had not been evaluated and that Exelon had not performed the timed test required by 10 CFR 50.63 to show that actions required to provide power to the blacked-out unit from the AAC could be performed within the analysis requirements. As a result, the team concluded that Exelon did not demonstrate that the AAC source would have the required availability and capability within the analyzed timeline. Exelon entered the issue into their corrective action program for evaluation and resolution.

  This issue was more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team determined the finding was of very low safety significance because it was a design or qualification deficiency confirmed not to result in a loss of functionality. The finding had a cross-cutting aspect in the area in the area of Problem Identification and Resolution, Corrective Action Program Component, because Exelon did not thoroughly evaluate problems such that resolutions address causes and extent of conditions and did not conduct effectiveness reviews to ensure problems are resolved. Specifically, Exelon's recent safety evaluation did not evaluate problems associated with a loss of an EDG due to a high temperature condition and the impact on the SBO AAC power source availability. (IMC 0310, Aspect P.1(c)) (1R17.1b)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (IP 71111.17)

.1 Evaluations of Changes, Tests, or Experiments (27 samples)

a. Inspection Scope

The team reviewed eight safety evaluations to determine whether the changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), had been reviewed and documented in accordance with 10 CFR 50.59 requirements. In addition, the team evaluated whether Exelon had been required to obtain NRC approval prior to implementing the changes. The team interviewed plant staff and reviewed supporting information including calculations, analyses, design change documentation, procedures, the UFSAR, the Technical Specifications (TS), and plant drawings to assess the adequacy of the safety evaluations. The team compared the safety evaluations and supporting documents to the guidance and methods provided in Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Evaluations," as endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," to determine the adequacy of the safety evaluations.

The team also reviewed a sample of nineteen 10 CFR 50.59 screenings for which Exelon had concluded that no safety evaluation was required. These reviews were performed to assess whether Exelon's threshold for performing safety evaluations was consistent with 10 CFR 50.59. The sample included design changes, calculations, and procedure changes.

The team reviewed the safety evaluations that Exelon had performed and approved during the time period covered by this inspection (i.e., since the last modifications inspection) not previously reviewed by NRC inspectors. The screenings and applicability determinations were selected based on the safety significance, risk significance, and complexity of the change to the facility.

In addition, the team compared Exelon's administrative procedures used to control the screening, preparation, review, and approval of safety evaluations to the guidance in NEI 96-07 to determine whether those procedures adequately implemented the requirements of 10 CFR 50.59. The reviewed safety evaluations and screenings are listed in the attachment.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50.63, "Loss of All Alternating Current Power," because Exelon did not demonstrate that the alternate alternating current (AAC) source provided the availability and capacity needed to mitigate a station blackout (SBO) within the analyzed one-hour coping timeline. Specifically, the team determined that Exelon's evaluation of the availability of the non-blackout unit's excess emergency diesel generator (EDG) capacity did not analyze the effects of the temporary loss of one of the three credited EDGs following an assumed single failure on the non-blacked out unit.

Description: The team reviewed Exelon's 10 CFR 50.59 safety evaluation that evaluated changes to the emergency service water (ESW) system valve configuration. This safety evaluation was performed following the issuance of unresolved item (URI) 05000352, 353/2011008-01, "Station Blackout Licensing Basis Assumed Alternate AC Power Source." The URI documented the need for further evaluations to determine if the AAC power source was able to meet Limerick Generating Station's (LGS) licensing basis during certain SBO events. The team reviewed the NRC Supplemental Safety Evaluation for Station Blackout Rule (10 CFR 50.63) for Limerick Units 1 and 2, dated June 10, 1992, which documented the NRC staff's evaluation of the LGS's loss of all alternating current power submittal. In the NRC Safety Evaluation, the staff approved the use of an AAC power source to supply alternating current (AC) power to the blacked-out unit. The team found that the NRC Safety Evaluation allowed the AAC source to be the excess capacity from the non-blackout unit's EDGs. The NRC Safety Evaluation concluded that with an assumed single failure of one of the four EDGs on the non-blackout unit, the remaining three EDGs were assumed to be available and had sufficient capacity to shutdown both units safely.

During a review of Exelon's 10 CFR 50.59 safety evaluation, the team found that, following the changes to the ESW lineup, one of the three credited EDGs would trip under certain SBO scenarios. Specifically, in the event of a Unit 1 SBO and the assumed single failure of an EDG on Unit 2 (the non-blacked out unit), ESW cooling to one of the remaining EDGs would be lost. This would result in a subsequent high jacket water temperature trip of the EDG. As a result, the non-blackout Unit 2 would have two operating EDGs and require operator action to recover the third EDG in order to provide the excess capacity (three non-blackout unit EDGs) assumed in the NRC Safety Evaluation. The team noted that Exelon had determined that power to the affected ESW pump would be restored by providing power to the SBO unit's 4kV bus from a non-blackout unit 4kV bus via a safety bus cross-connection in accordance with the SBO emergency operation procedures. Exelon concluded that once power was restored to an ESW pump, ESW flow would be restored allowing for recovery of the third EDG, therefore, no change to the license was required.

The team identified that Exelon's evaluation of the loss of the third EDG due to a high jacket water temperature trip did not consider the time required for the temperature trip to reset. The team found that Exelon had assumed that when ESW was restored to the EDG the temperature switch would quickly reset. However, because the jacket water pump would not be operating the team questioned how long it would take to cool the jacket water system in order to reset the trip and allow the EDG to be started. In response to the teams questions, Exelon performed thermal calculations to determine the time required to cool down sufficiently; however, because there was a large variance in the time based on calculation assumptions the team concluded that the calculations did not demonstrate that the third EDG could be restored within the analyzed one-hour coping timeline. Additionally, the team could not determine the actual time available to allow for recovery of the EDG because Exelon did not have records of a demonstration that recorded the time required to power the blacked out unit from the AAC source. Therefore, the team concluded the Limerick Generating Station's AAC power was not in conformance with the analysis and did not meet SBO requirements. The team noted that although LGS was not in conformance with the analysis submitted to the NRC to demonstrate compliance with the SBO Rule, LGS did have procedures in place and additional equipment capacity (EDG and DC battery) not credited in the analysis that would allow the unit to cope with a station blackout until the third EDG could be restarted. Exelon entered these issues into their corrective action program for evaluation and resolution under CR 01288965.

Analysis: The team determined that the failure to verify the AAC source would be available within the analyzed timeframe during an SBO event was a performance deficiency. Specifically, Exelon's 10 CFR 50.59 safety evaluation did not include a complete evaluation of the affects of a high temperature trip on a non-blackout EDG with respect to the non-blackout unit's ability to provide the AAC source within the analysis timeline assumed in the NRC Safety Evaluation. The team concluded that this performance deficiency was reasonably within Exelon's ability to foresee and prevent. This issue was more than minor because it was similar to NRC Inspection Manual Chapter (IMC) 0612, Appendix E, "Examples of Minor Issues," Example 3.j, in that as a result of this deficiency; the team had a reasonable doubt of operability with respect to the AAC power source capacity to recover from an SBO. In addition, the finding was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, capability, and reliability of systems that respond to initiating events to prevent undesirable consequences.

The team performed a Phase 1 SDP screening, in accordance with NRC IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," and determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of functionality of the equipment. Specifically, a single failure of an EDG on the non-blacked out unit did not need to be assumed per the SDP. The team identified a cross-cutting aspect associated with the finding in the area of Problem Identification and Resolution, Corrective Action Program Component, because Exelon did not thoroughly evaluate problems such that resolutions address causes and extent-of-conditions and did not conduct effectiveness reviews to ensure problems are resolved. Specifically, Exelon's recent 10 CFR 50.59 safety evaluation did not evaluate problems associated with a loss of an EDG due to a high temperature condition and the impact on the SBO AAC availability. (IMC 0310, Aspect P.1(c))

Enforcement. 10 CFR 50.63, "Loss of All Alternating Current Power," requires that a plant be able to withstand for a specified duration and recover from an SBO. An AAC power source constitutes the acceptable capability to withstand an SBO provided an analysis is performed which demonstrates that the plant has this capability from onset of the SBO until the AAC source and required shutdown equipment are started and lined up to operate. In addition, the time required for startup and alignment of the AAC power source and this equipment shall be demonstrated by test. Contrary to the above, after changes were made to the ESW lineup in October 12, 2001, Exelon did not determine whether the non-blackout unit's EDGs were capable of providing the necessary excess capacity within the analyzed one hour coping timeframe. Because this finding was of very low safety significance and was entered into the corrective action program (IR 01288965), this violation was treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000352/2011007-01, Failure to Verify Alternate AC Source Capability to Recover from Station Blackout)**

.2 Permanent Plant Modifications (10 samples)

.2.1 Use of Ultra-Low Sulfur Diesel Fuel for the Emergency Diesel Generators

a. Inspection Scope

The team reviewed a modification (07-00049) that evaluated the acceptability of transitioning from S500 (500 ppm sulfur) low sulfur diesel fuel oil to S15 (15 ppm sulfur) ultra-low sulfur diesel (ULSD) fuel oil for use in the EDGs. The transition was made to meet Environmental Protection Agency rules and standards. Exelon performed the modification in order to evaluate the effect ULSD fuel oil would have on the performance capability of the EDGs, and to verify that the design and licensing bases for LGS were not impacted by the use of the ULSD fuel oil. Additionally, the evaluation determined the actions required by the site to support the fuel change.

The team reviewed Exelon's evaluations for use of the ULSD fuel oil, as well industry operating experience, to determine if compatibility issues with ULSD fuel oil were appropriately addressed. The team reviewed the revised diesel storage and fuel oil consumption calculations, and discussed the calculations with the responsible design engineers to determine if the calculation assumptions were appropriate and the required volume of ULSD fuel oil was in accordance with the licensing requirements of the plant. The team also reviewed fuel oil procurement and sample procedures, and receipt records to determine if Exelon was appropriately monitoring ULSD fuel oil parameters. Finally, the team reviewed condition reports (CRs) and EDG testing records to verify that EDG performance was not impacted by the fuel oil change. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. Documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.2 Modification to the U1 High Pressure Coolant Injection Booster Pump Coupling

a. Inspection Scope

The team reviewed a modification (10-00126) to the high pressure coolant injection (HPCI) booster pump coupling. The coupling connects the rotating assemblies of the HPCI booster pump and the HPCI main pump. Exelon performed this modification to allow for improved and easier disassembly of the coupling for maintenance activities.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of the HPCI pump had not been degraded by the modification. The team interviewed Exelon's engineering staff and reviewed the vendor technical evaluation to determine if the coupling modification had impacted the pump or coupling performance. The associated work order instructions and documentation were reviewed to verify that maintenance personnel implemented the modification as designed. The team also walked down the HPCI booster pump and HPCI booster pump coupling to determine if the maintenance activities were performed in accordance with the modification procedures. Finally, the team reviewed surveillance test results to determine if the HPCI pump performance had been adversely impacted. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. Documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.3 Replacement of the 2B-E205 Residual Heat Removal Heat Exchanger

a. Inspection Scope

The team reviewed a modification (09-00333) that replaced the 2B-E205 residual heat removal (RHR) heat exchanger. The RHR heat exchanger removes heat from the reactor after plant shutdown, and removes heat from the primary containment during certain design basis accidents. Exelon implemented this modification to replace the existing RHR heat exchanger that was approaching design limits. The new heat exchanger was selected by Exelon because it was dimensionally similar to the existing heat exchanger, and was built with alloy steel tubes which had improved corrosion resistance and performance for the component.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of the RHR heat exchanger and associated system had not been degraded by the material change to the heat exchanger tubes or the heat exchanger installation. The team interviewed design engineers and reviewed vendor data, calculations, and evaluations to determine if the capacity of the new heat exchanger met the design and licensing requirements. Additionally, the team reviewed post-modification testing (PMT) results, and associated maintenance work orders to verify that the heat exchanger replacement modification was appropriately implemented.

Finally, the team walked down the heat exchanger with the system engineer to verify the maintenance activities were performed as described in the modification package. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. Documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.4 Modification of Residual Heat Removal Service Water 'B' Return Loop Piping

a. Inspection Scope

The team reviewed a modification (09-00134) that installed a drain valve assembly in the RHR service water (RHRSW) return loop piping. The RHRSW return loop piping returns RHRSW to the spray pond. Exelon performed the installation of the drain valve assembly to repair a pipe flaw.

The team reviewed the modification to determine if the design basis, licensing basis, or performance capability of the return line had been degraded by the modification. The team interviewed design and non-destructive testing engineers, and reviewed evaluations, non-destructive testing results, PMT results, and associated maintenance work orders. This review was performed to verify the flaw was repaired by the installation of the assembly, the repair met the requirements of the American Society of Mechanical Engineers (ASME) Code, and that the drain valve modification was appropriately implemented. The team also verified that the drain valve assembly specifications, associated procedures, and drawings had been updated. Finally, the team walked down the drain valve with the system engineer to verify the maintenance activities were performed in accordance with the work order. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. Documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.5 Unit 2 Measurement Uncertainty Recapture Power Uprate Leading Edge Flow Meter

a. Inspection Scope

The team reviewed a modification (09-00097) that installed the Leading Edge Flow Meter (LEFM) CheckPlus System in Unit 2's three main feedwater piping return headers. The modification was performed to reduce the two percent uncertainty margin as originally required by 10 CFR Part 50, Appendix K. Feedwater flow signals from installed flow venturis had been used for determining core thermal power. The LEFM modification was performed to provide feedwater mass flow signals as the primary input to determine core thermal power. The modification included installation of a metering spool piece that consisted of 16 ultrasonic transducers, a common pressure tap for two new pressure transmitters, and a thermowell for the dual element resistance temperature detector.

The team reviewed the modification to determine if the design basis, licensing basis, and performance capability of the feedwater flow measurement system had been degraded by the modification. The team reviewed calculations and technical evaluations, and interviewed system and design engineers to assess whether the modification was consistent with design assumptions. Replacement components and materials were reviewed to ensure that the modification conformed to the design specifications for the feedwater system. The team also reviewed design assumptions and supporting uncertainty calculations to evaluate whether they were technically appropriate and consistent with the UFSAR, and to ensure design limits were not exceeded. The team reviewed the post-modification testing and vendor commissioning documents to verify proper operation of the system. Finally, the team reviewed CRs associated with the system installation to verify that deficiencies were appropriately identified and corrected. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. Documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.6 Technical Specification 3.8.1 Intent Changed Without Prior NRC Approval

a. Inspection Scope

The team reviewed a modification (09-00284) that returned wording in the TS Bases 3/4.8.1 document to the wording used prior to implementation of modification 99-00682. In 1999, Exelon implemented 99-00682 which changed the TS Bases 3/4.8.1 wording to state that only three out of four 4 kV emergency buses were required to be electrically connected to offsite power to maintain the operability of the offsite power sources. In a previous inspection report, NRC inspectors determined that this change to the TS Bases changed the intent of the associated TS 3/4.8.1 and issued a Severity Level IV non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," (NCV 05000352, 353/2009002-02) for failing to obtain a TS license amendment prior to changing the wording. Modification 09-00284 changed the TS Bases wording to require all four emergency buses be connected to offsite power in order to consider offsite power to be operable.

The team reviewed the modification to verify that the approved design and licensing bases had not been changed by the modification. The team noted the modification was only a change to the TS Bases document and did not require any plant equipment changes. The team reviewed the design and licensing bases assumptions to evaluate whether the modification was appropriate and consistent with the UFSAR. Also, the team reviewed CRs associated with the original modification and associated violation to determine if the deficiencies identified were appropriately corrected. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. Documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.7 Multiple Spurious Operation: Generate 2R11 ECR for Mods to Core Spray and Residual Heat Removal Check Valves

a. Inspection Scope

The team reviewed a modification (10-00347) that changed the test circuit wiring for the core spray (CS) and RHR testable check valves in order to prevent the valves from spuriously opening, due to a hot short, during a postulated fire scenario. The modification was performed because a hot short could bypass the testable check valve pushbutton switches in the main control room and cause the testable check valves to spuriously open. The modification added wiring to the test circuit of each testable check valve in order to cause a short to ground in the event that a postulated hot short occurred during a fire.

The team reviewed the design basis, licensing basis, and performance capability of the CS and RHR testable check valves. The team evaluated the modification to ensure it was consistent with requirements in the design and licensing bases, and that the components had not been degraded. The team reviewed technical evaluations to determine whether the modification was consistent with design assumptions for valve operation. Electrical elementary wiring diagrams were reviewed to verify that the testable check valves were not adversely affected by the modification, and replacement materials were reviewed to ensure that they conformed to the system design specifications. The team also verified selected drawings and procedures were properly updated for the new equipment configuration. Additionally, the team reviewed the post-modification testing performed to verify proper operation of the CS and RHR testable check valves to determine if the results were satisfactory. Finally, the team conducted interviews with engineering staff to determine if the testable check valves functioned in accordance with the design assumptions, and if the modification corrected the previously identified problem. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. Documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.8 Incorporated Shroud Evaluation into Design Basis

a. Inspection Scope

The team reviewed a modification (09-00035) which was an engineering evaluation performed to re-evaluate the structural integrity of the Unit 2 core shroud. Specifically, the modification evaluated the welds on the core shroud to determine an acceptable time interval for in-service inspections of the welds that connected the sections of the shroud. The evaluation determined how many cycles of operation could occur before re-inspection of the core shroud welds would be required to validate the assumptions in the methodology used to determine the structural integrity of the weld. To perform this evaluation Exelon utilized the RAMA (Radiation Analysis Modeling Application) Code methodology.

The team reviewed the modification to determine if the design and licensing bases requirements for the Unit 2 core shroud welds were met. The team assessed if the methodology was in accordance with the guidance of Regulatory Guide 1.190 and evaluated the basis for the inputs into the code. The team also determined if Exelon satisfactorily evaluated the results of the evaluation in order to determine the appropriate shroud weld inspection interval. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.9 Total Integrated Dose (TID) Evaluation for Drywell Coatings (paint)

a. Inspection Scope

The team reviewed a modification (11-00122) which revised calculation LM-0675 - TID Evaluation for Drywell Coatings. The calculation determined the total dose to qualified coatings inside the drywell. For the new calculation, Exelon changed the evaluation methodology from an infinite cloud evaluation to the semi-infinite cloud model because it was determined that the infinite cloud model overestimated the total dose to the coatings. Additionally, the revision to the calculation was based on a 60 year expected dose to the coatings.

The team reviewed the modification to determine if the design and licensing bases for the evaluation of the drywell coating systems remained valid. The team reviewed the calculation to verify the assumptions used were valid and the coatings had been qualified to receive the doses determined by the calculation without failing. Finally, the team determined if the new methodology was an acceptable methodology for determining coating dose and had been reviewed by the NRC. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.10 Setpoint Change for Temperature Indicating Switch (TIS)-025-101/201

a. Inspection Scope

The team reviewed modification (09-00551) which evaluated the permanent change to the TIS-025-101/201. The TIS actuates based on the delta-temperature (delta-T) HPCI room trip set-point. The delta-T trip provides a signal to isolate the HPCI steam piping in the event of a design basis steam leak in the room. The set-point change was made because the previous set-point was determined to be non-conservative. During revisions of various calculations, Exelon determined that room temperature following a steam line break would not exceed the previous trip set-point.

The team reviewed the modification to verify that the design and licensing bases of the isolation system had not been degraded by the set-point modification. The team determined if Exelon had evaluated the impact of the delta-T trip set-point and appropriately calculated the new set-points. The team also verified the calibration procedures were updated for the revised set-points. Finally, the team reviewed the technical specifications to verify that limits in the TS had been appropriately revised and that no TS violations had occurred. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed a sample of CRs associated with 10 CFR 50.59 and plant modification issues to determine whether Exelon was appropriately identifying, characterizing, and correcting problems associated with these areas, and whether the planned or completed corrective actions were appropriate. In addition, the team reviewed CRs written on issues identified during the inspection to verify adequate problem identification and incorporation of the problem into the corrective action system. The CRs reviewed are listed in the attachment.

b. Findings

No findings were identified.

4OA4 Other

a. Unresolved Item 05000352,353/2011008-01 - Station Blackout Licensing Basis Assumed Alternate AC Power Source (Closed)

The team reviewed URI 05000352,353/2011008-01, "Station Blackout Licensing Basis Assumed Alternate AC Power Source." The URI was opened to evaluate if the changes that Exelon performed on the ESW system lineup impacted the SBO licensing basis. Specifically, NRC inspectors determined that following a worst case single failure on the non-blacked out unit (including the single failure of the EDG assumed in the NRC Safety Evaluation on SBO), the third EDG credited in the SBO analysis would trip on high temperature and questioned whether this would be considered a malfunction of the EDG and, therefore, the EDG could not be credited under the current licensing basis. Exelon acknowledged that the EDG may trip on high temperature but believed that the EDG could be recovered and, therefore, be credited as one of the three EDGs required by the licensing basis for SBO.

The team reviewed the NRC Safety Evaluation for Limerick Generating Station, dated June 10, 1992, to determine the EDG configurations required to mitigate an SBO. The team determined that the NRC's approval of Exelon's strategy to meet the SBO rule was based on excess capacity from the non-blacked out unit's EDGs. During a single unit SBO the non-blacked out unit was assumed to have three of their four EDGs available. This scenario was based on a common cause failure of all EDGs on the blacked out unit and an assumed single active failure of one EDG on the non-blacked out unit. The team found that the non-blacked out unit required the capacity of more than one but less than two EDGs to achieve safe shutdown. The analysis credited the excess capacity of the three remaining EDGs to be available to safely shutdown the unit affected by the station blackout, during the four hour coping period of the SBO. Additionally, the team determined that the NRC Safety Evaluation stated that the excess capacity would be available within one hour of the start of the SBO.

The team reviewed Exelon's SBO procedures, electrical configurations, and emergency service water alignments to determine if the systems were able to be cross-tied, if plant procedures correctly directed operators to complete the alignment, and if the excess capacity would be available within one hour. Specifically, the team reviewed the event scenario where the third EDG tripped due to the worst case single failure and operator action was required to recover the EDG. The team concluded that if the excess capacity was able to be placed on the blacked out unit's vital buses within one hour, the LGS licensing basis was met. This unresolved item is closed.

b. Findings

One finding was identified. See Section 1R17.1.b. for details.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. Peter Gardner, Plant Manager, and other members of Exelon's staff at an exit meeting on November 4, 2011. The team returned the proprietary information reviewed during the inspection and verified that this report does not contain proprietary information.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee Personnel</u>

P. Gardner	Plant Manager
D. Doran	Director of Engineering
W. Lewis	Senior Manager Engineering Design
R. George	Manager Electrical Design
R. Harding	Regulatory Assurance
A. Lambert	Design Engineer
J. Mittura	Design Engineer
M. Gift	Design Engineer
K. Collier	Design Engineer
N. Roy	Design Engineer
R. Schwab	Design Engineer
E. Hosterman	Design Engineer
L. Hemler	System Engineer
J. Berg	System Engineer

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

<u>Opened and Closed</u>

05000352/2011007-01	NCV	Failure to Evaluate Station Blackout Timeline for EDG Availability (section 1R17.1b)

<u>Closed</u>

05000352,353/2011008-01	URI	Station Blackout Licensing Basis Assumed Alternate AC Power Source (section 4OA5)

LIST OF DOCUMENTS REVIEWED

<u>10 CFR 50.59 Evaluations</u>

LG2009E001, 'C' SLCS Pump Control Switch Modification to Inhibit Automatic Pump Start on Units 1 and 2, Rev. 0
LG2009E002, Provide an Alternate Means of Monitoring Reactor Well Seals Leak, Rev. 1
LG2010E002, Suppression Pool Gross Input Leak Rate Determination, Unit 1 Rev. 15 and Unit 2 Rev. 18
LG2010E003, Modify Select MOV Circuits to Prevent Spurious Operations during Postulated Hot Short Fire Scenarios, Rev. 0
LG2011E001, ECR LG 10-00103 use of TRACG04P Version 4.2.60.3, Rev. 0
LG2011E002, Modify Select MOV Circuits to Prevent Spurious Operations during Postulated Hot Short Fire Scenarios, Rev. 0

LG2011E003, Reactor Recirculation M/G Set Replacement with Adjustable Speed Drive (ASD) Units, Rev. 0
LG2011E004, Assessment of the Affect of ECR 01-00816 on Station Black Out Coping, Rev. 0

10 CFR 50.59 Screened-out Evaluations
LG2009S004, Incorporating Shroud Evaluation into Design Basis, Rev. 0
LG2009S010, UFSAR Chapter 8 Changes - Spare Safeguard Transformer, Rev. 0
LG2009S025, Revised Calcs M-81-10, M-81-27 and M-81-28, Rev. 0
LG2009S026, UFSAR Section 8.1 Revision (Offsite Sources), Rev. 0
LG2009S032, Tech Spec 3.8.1 Intent Changed without Prior NRC Approval, Rev. 0
LG2009S036, ESW Loop 'A' Flow Balance, Rev. 0
LG2009S054, Permanent Setpoint Change for TIS-025-101/201 B&D, Rev. 1
LG2010S016, New Allowable Total Connection Resistances for Station Batteries, Revs. 36 and 38
LG2010S066, Leading Edge Flow Meter CheckPlus Installation, Rev. 1
LG2010S073, Installation of Support Equipment for CRE Connections in Unit 2 AC/BD RHR Rooms, Rev. 0
LG2011S001, Fluence Calculation Incorporation, Rev. 0
LG2011S022, MSOPS: Generate ECR for 1R14 DC Bucket Mods, Rev. 0
LG2011S028, LGS RHR and Core Spray Loop 'A' Unit 1 Testable Check Valve/Bypass Valve Circuit Modification, Rev. 0
LG2011S035, Prepare ECR for Revision of TID Calc for Drywell Coatings, Rev. 0

Modification Packages
01-00816, Operability Determination and NCR for ESW and Emergency D/G's, Rev. 0
07-00049, Use of Ultra-Low Sulfur Diesel Fuel, Rev. 2*
09-00035, Incorporated Shroud Evaluation into Design Basis, Rev. 0
09-00097, Unit 2 Measurement Uncertainty Recapture (MUR) Power Uprate Leading Edge Flow Meter (LEFM), Rev. 1
09-00134, HBC-507-01 SW2 Piping Modification, Rev. 2*
09-00284, Tech Spec 3.8.1.1 Intent Changed Without Prior NRC Approval, Rev. 0*
09-00333, Replacement of 2B RHR Heat Exchanger, Rev. 8*
09-00485, Compartment Temperature Transients for Steam and Water Leaks, Rev. 0
09-00551, Permanent Setpoint Change for TIS-025-101/201 B&D, Rev. 2
10-00126, HPCI Booster Pump Coupling Modification, Rev. 0*
10-00347, Multiple Spurious Operations: Generate 2R11 ECR for Mods to Core Spray and Residual Heat Removal Check Valves, Rev. 2333
11-00122, TID Evaluation for Drywell Coatings, Rev. 0
(* designates a Modification and 10 CFR 50.59 screen-out evaluation sample)

Calculations, Analysis, and Evaluations
0000-0125-5142, HPCI Speed Increase Evaluation, Rev. 0
364586, Ultra-Low Sulfur Diesel Fuel Evaluation, dated 2/16/07
6380E.07, Diesel Generator Loading (Steady State), Rev. 12
ER-LG-331, Augmented Inspection Program – Aug 20 Core Shroud Inspection, Rev. 1
HBC-507-H002, Temporary Brace at Pipe Support, Rev. 0
LE-0052, Class 1E Battery Load Duty Cycle Determination, Rev. 12
LE-0111, Xformer Inrush and Motor Starting Current Transients during EDG Cross-Tie, Rev. 0

LE-0114, Reactor Core Thermal Power Uncertainty Calculation Unit 2, Rev. 1
LEAE-MUR-0003, Bounding Uncertainty Analysis for Thermal Power Determination at Limerick Unit 2 using LEFM CheckPlus System, Rev. 0
LG-MISC-02, PRA Sensitivity Study for the Potential Impacts of Increasing the Suppression Pool Cooling Run Time, Rev. 0
LG-PRA-010, LGS PRA Data Notebook Volume 1, Rev. 1
LM-0007, Diesel Generator Fuel Oil Consumption, Rev. 4
LM-0052, Differential Pressure Calculations for MOVs in the HPCI System, Rev. 7
LM-014, Determine Sizing and Configuration of LGS Unit 1 RHR Test Return Line, Rev. 1
LM-0663, Diesel Generator Day Tank Minimum Level, Rev. 2
LM-0675, TID Evaluation for Drywell Coatings, Rev. 0
LM-280, Radiation Through Bioshield Wall and Streaming Through Bioshield Penetration, dated 3/2/93
M-11-32, Heat Exchangers Input Data for Computer Performance Program, Rev. 5
M-55-03, HPCI Steam Supply Pressure Drop, Rev. 6
M-55-20, HPCI Pump Discharge Maximum Pressure, Rev. 5
M-81-10, Spray Pond Pump Facility Ventilation Requirements, Rev. 4
M-81-27, Spray Pond Pump Station - Minimum Temperature in the Small Room, Rev. 3
M-81-28, Spray Pond Pump Structure Temperature-Time Curve After a LOCA/LOOP, Rev. 2
NED C-32847P, ARTS Flow-Dependent Limits with Turbine Bypass Valve Out of Service for Peach Bottom Atomic Power Station and Limerick Generating Station, dated 6/98

Condition Reports

00534749	00723472	01138861	01286023*
00656269	00885528	01282425	01286047*
00673832	00905220	01285226*	01288965*
00691575	01047576	01285263*	

(* denotes NRC identified during this inspection)

Drawings
8031-M-11, Sht. 1, Emergency Service Water, Rev. 70
8031-M-12, Shts. 1-2, Residual Heat Removal Service Water, Revs. 70 and 7
8031-M-51, Shts. 1-8, Residual Heat Removal, Revs. 65, 66, 67, 66, 30, 23, 21, and 25
8031-M-53, Sht. 3, P&ID Fuel Pool Cooling and Cleanup, Rev. 16
8031-M-56, HPCI Pump/Turbine Unit 1, Rev. 40
CA34471, Forged Steel Maximum Bore Hub Puller Holes, Rev. 2
M-1-E11-1040-E-032, Sh. 1, Elementary Diagram Residual Heat Removal System, Rev. 13
M-1-E11-1040-E-035, Sh. 1, Elementary Diagram Residual Heat Removal System, Rev. 8

Procedures
A-C-134, Control of Hazards Barriers, Rev. 4
ARC-BOP-20C222, D3 Reactor Well Seal, Rev. 0
ARC-MCR-107, A-3 Alarm Response Card, Rev. 1
ARC-MCR-212, I5 Fuel Pool Storage Hi/Lo Level, Rev. 1
E-1, Loss of All AC Power (Station Blackout), Rev. 40
E-10/20, Loss of Offsite Power, Rev. 44
ER-AA-340, Generic Letter 89-13 Program Implementing Procedure, Rev. 6

IC-11.00388, Calibration of HPCI Turbine Governor Control System for the Limerick Generating Station, Rev. 8
LS-AA-104, Exelon 50.59 Review Process, Rev. 6
M-093-004, 480 VAC MCC Breaker Assembly and Cubicle Terminal Maintenance, Rev. 10
OS12.1.A, Alignment for Normal Operation of the Residual Heat Removal Service Water System for Loop 'B', Rev. 20
P-305, Welding and Non-Destructive Testing Requirements for Field Erected Piping, Rev. 27
PES-P-006, Diesel Fuel Oil, Rev. 8
RT-2-012-391-2, 2B-E-205 RHR Heat Exchanger Transfer Test, Rev. 6
RT-6-041-490-1, Suppression Pool Gross Input Leak Rate Determination, Rev. 16
RT-6-041-490-2, Suppression Pool Gross Input Leak Rate Determination, Rev. 19
S51.8.A, Suppression Pool Cooling Operation and Level Control, Rev. 42
S53.0.A, Normal Makeup/Response to Low Level in Fuel Storage Pool or Reactor Well, Rev. 22
S92.3.N, Receiving Diesel Fuel Oil Delivery, Rev. 36
ST-5-020-810-0, Diesel Generator Fuel Oil Receipt Analysis, Rev. 28
ST-5-020-811-1, Diesel Generator Fuel Oil Post Receipt Analysis, Rev. 14
ST-6-055-230-1/2, HPCI Pump, Valve, and Flow Test, Rev. 76

Work Orders

C0231554	C0235924	R1030435	R1101708
C0234023	R1010379	R1069699	R1108012
C0235918	R1023526	R1075525	R1141314
C0235919			

Miscellaneous
ANSI N101.2, Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities, dated 1972
ASTM D4082-10, Standard Test Method for Effects of Gamma Radiation on Coatings for Use in Nuclear Power Plants
BWRVIP-114-A, BWR Vessel and Internals Project – RAMA Fluence Methodology Theory Manual, dated 6/09
GE-NE-0000-0052-5690, TRACG04 DIVOM 10 CFR 50.59 Evaluation Basis, Rev. 0
GE-NE-0000-0115-7421, TRACG04P (Version 4.2.60.3) DIVOM 10 CFR 50.59 Evaluation Basis, Rev. 0
GNF-S-0000-0109-4007, TRACG04P Error Correction 10 CFR 50.59 Evaluation Basis, Rev. 1
NEI 96-07, Nuclear Energy Institute Guidelines for 10 CFR 50.59 Implementation, Rev. 1
NRC Generic Letter 1998-13, Service Water System Problems Affecting Safety-Related Equipment, dated 7/18/89
NRC Information Notice 1987-10, Potential for Water Hammer during Restart of Residual Heat Removal Pumps, dated 2/11/87
NRC Information Notice 2006-22, New Ultra-Low-Sulfur Diesel Fuel Oil could Adversely Impact Diesel Engine Performance, dated 10/12/06
NRC Information Notice 2010-17, Common Cause Failure of BWR Recirculation Pumps with Variable Speed Drives, dated 9/10/10
NUMARC 87-00, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors, Rev. 1
NUREG-0588, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, Rev. 1

TP24, Caldon Ultrasonics Verification and Validation Data Package Documents Vol. II, Rev. 21
TSH-GA-110A-1, Instrument Calibration Sheet

Surveillance and Modification Acceptance Tests

0000-0129-8688-R1, Summary of GEH Transient Anticipated Operational Occurrences (AOO) with Respect to ASD Modification in LGS Units 1 and 2, Rev. 1
A5E02029143A, High Availability VFD Drive: Failure Modes Effects Analysis and Probabilistic Risk Assessment, Rev. AE
ER-790, An Evaluation of the Impact of 55 Tube Permutit Flow Conditions on the Meter Factor of an LEFM CheckPlus System, Rev. 1
ER-797, Meter Factor Calculation and Accuracy Assessment for Limerick Unit 2, Rev. 0
FCDP-197, LEFM CheckPlus 2000FC Flow Measurement System Field Commissioning Data Package, Rev. 0
MAT 09-00097-1, Unit 2 LEFM Modification Acceptance Test, Rev. 0
ST-4-015-490-2, Reactor Well Seals Leak Test, performed 3/22/09
ST-5-020-810-0, Diesel Generator Fuel Oil Receipt Analysis, performed 8/9/11 and 8/19/11
ST-5-020-811-1, Diesel Generator Fuel Oil Post Receipt Analysis, performed 10/6/10 and 8/11/11
ST-6-051-232-2, 'B' RHR Pump, Valve, and Flow Test, performed 4/19/11
ST-6-092-111-1, Diesel Generator 24-Hour Endurance Test, performed 9/29/10

Design & Licensing Bases

Letter from Philadelphia Electric Company to NRC, Limerick Generating Station, Units 1 and 2 10 CFR 50.63, "Loss of All Alternating Current Power" Supplemental Information, dated 4/9/90
Letter from Philadelphia Electric Company to NRC, Limerick Generating Station, Units 1 and 2 10 CFR 50.63, "Loss of All Alternating Current Power" Response to NRC Safety Evaluation, dated 9/4/91
Letter from USNRC to EPRI, US Nuclear Regulatory Commission Approval Letter for BWRVIP-117-A, "RAMA Fluence Methodology for Plant Application - Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5", dated 4/18/11
Limerick Generating Station Units 1 and 2 - Shutdown of the Non Blacked-out Unit with 2 Diesels in the First Hour Following an SBO, dated 10/21/2011
Limerick Generating Station Updated Final Safety Accident Report, Rev. 15
L-S-07, Diesel Generator and Auxiliary Systems DBD, Rev. 12
Safety Evaluation by The Office of Nuclear Reactor Regulation - Station Blackout Safety Evaluation Philadelphia Electric Company, Limerick Generating Station Units 1 and 2, dated 6/10/92
SAIC-91/6651, Technical Evaluation Report Limerick Generating Station, Units 1 and 2 Station Blackout Evaluation, dated 3/8/91

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AAC	Alternate Alternating Current
AC	Alternating Current
ASD	Adjustable Speed Drive
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR	Condition Reports
CS	Core Spray
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
LEFM	Leading Edge Flow Meter
LGS	Limerick Generating Station
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PMT	Post Modification Test
ppm	Parts Per Million
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
SBO	Station Blackout
SDP	Significance Determination Process
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
ULSD	Ultra Low Sulfur Diesel
URI	Un-resolved Item